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4.9.0 Shutdown Plant Problems

Learning Objectives:

1. State the purposes of the residual heat removal (RHR) system.
2. Describe the alignment and operation of the RHR system during its shutdown cooling mode of operation.
3. Describe design features of the RHR system which could reduce its reliability when it is being used for decay heat removal.
4. Describe the consequences of losing decay heat removal capability when the reactor is in cold shutdown.

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4.9.1 Introduction

One of the most significant problems associated with a shutdown reactor is the removal of the heat being produced by radioactive decay of the fission products produced during reactor operation. The General Design Criteria in Appendix A of 10CFR50 address this problem by requiring a residual heat removal system to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capability shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The core damage frequency associated with the inability to remove decay heat from the reactor core was demonstrated to be significant in the results of the Reactor Safety Study (WASH-1400). The overall probability of core damage in the first generation of large commercial power reactors was higher than had been expected (about 5×10^{-5} as compared to 1×10^{-6} per reactor year). Inadequate reliability of the decay heat removal system (specifically following a small-break loss of coolant accident) was shown to be responsible for a substantial portion of the overall probability of core damage. This fact, combined with repetitive events resulting in the inadequate or complete loss of decay heat removal capability in operating plants, led the NRC to designate shutdown decay heat removal requirements as an Unresolved Safety Issue (USI A-45). Under the established task action plan, the NRC has studied the adequacy of systems for safely removing decay heat from a reactor core during shutdown and to assess the value and the impact of alternative measures for improving the reliability of the decay heat removal function.

4.9.2 RHR System Description

The purposes of the residual heat removal system are as follows:

1. Removes decay heat from the core and reduces the temperature of the reactor coolant system (RCS) during the second phase of plant cooldown,
2. Serves as the low pressure injection portion of the emergency core cooling systems (ECCSs) following a loss of coolant accident, and
3. Transfers refueling water between the refueling water storage tank and the refueling cavity before and after refueling.

The RHR system transfers heat from the reactor coolant system to the component cooling water system. During shutdown plant operations, the RHR system is used to remove the decay heat from the core and to reduce the temperature of the reactor coolant to the cold shutdown temperature (less than 200°F). The cooldown performed by the RHR system (from 350°F to less than 200°F), is referred to as the second phase of cooldown. The first phase of cooldown is accomplished by the auxiliary feedwater (AFW) system, the steam dump system, and the steam generators.

Once the plant is in cold shutdown, the RHR system will maintain RCS temperature until the plant is started up again. The residual heat removal system also serves as part of the emergency core cooling system during the injection and recirculation phases of a loss of coolant accident. The residual heat removal system is used to transfer refueling water between the refueling water storage tank and the refueling cavity before and after the refueling operations.

The residual heat removal system, as shown in Figure 4.9-1, consists of two heat exchangers, two residual heat removal pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet line to the residual heat removal system for the second phase of cooldown is connected to the hot leg of reactor coolant loop 4, and the return lines are connected to each cold leg of the reactor coolant system. These return lines also function as the emergency core cooling system low pressure injection lines.

The RHR pump suction line from the reactor coolant system is normally isolated by two series motor-operated valves (8701 and 8702). The suction line has a relief valve located downstream of the isolation valves; all three valves are located inside the containment. Each RHR supply to the RCS cold legs is isolated from the reactor coolant system by two check valves located inside the containment, and each RHR pump discharge line is can be isolated by a normally open motor-operated valve (8809A or 8809B) located outside the containment. These motor-operated valves are part of the emergency core cooling system and receive confirmatory open signals from the engineered safety features actuation system. During the second phase of cooldown, reactor coolant flows from the RCS to the residual heat removal pumps, through the tube side of the RHR heat exchangers, and back to the RCS. The heat from the reactor coolant is transferred to the component cooling water, which is circulating through the shell side of the RHR heat exchangers.

If one of the two pumps or one of the two heat exchangers is not operable, the ability to safely cool down the plant is not compromised; however, the time required for the cooldown is extended. The water chemistry requirements for the residual heat removal system are the same as those for the reactor coolant system. Provisions are made for extracting samples from the flow of reactor coolant downstream of the RHR heat exchangers for analysis. A local sampling point is also provided in each residual heat removal train between the pump and its associated heat exchanger.

To ensure the reliability of the RHR system, the two residual heat removal pumps are powered from separate vital electrical power supplies. If a loss of offsite power occurs, each vital bus is automatically transferred to a separate emergency diesel power supply. A prolonged loss of offsite power would not adversely affect the operation of the residual heat removal system.

The residual heat removal system is normally aligned to perform its safety function. Therefore, no valves are required to change position. For the RHR system to perform its safety function, the RHR pumps must start when the engineered safety features actuation signal is received, and the pressure in the reactor coolant system must drop below the discharge pressure of the RHR pumps.

The materials used to fabricate the RHR system components are in accordance with the applicable ASME code requirements. All parts or components in contact with borated water are fabricated of or clad with austenitic stainless steel or an equivalent corrosion-resistant material.

4.9.2.1 Component Description

4.9.2.1.1 Residual Heat Removal Pumps

Two pumps are installed in the residual heat removal system. The pumps are vertical, centrifugal units with mechanical seals on the shafts. These seals can be cooled by either component cooling water or service water, depending on the plant design. All pump surfaces in contact with reactor coolant are manufactured from austenitic stainless steel or an equivalent corrosion-resistant material. The pumps are sized to deliver reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements.

The residual heat removal pumps are protected from overheating and loss of suction flow by minimum flow bypass lines that assure flow to the pump suctions for pump cooling. A control valve located in each minimum flow line (610 or 611) is regulated by a signal from the flow transmitter located in each pump discharge header. Each control valve opens when the RHR pump discharge flow is less than 500 gpm and the pump is running, and closes when the flow exceeds 1000 gpm or the pump is not running. A pressure sensor in each pump header provides a signal for an indicator on the main control board. A high pressure annunciator alarm is also actuated by the pressure sensor.

4.9.2.1.2 Residual Heat Removal Heat Exchangers

Two heat exchangers are installed in the system. The heat exchanger design is based on the heat load and the temperature difference between the reactor coolant and component cooling water 20 hours after the reactor has been shut down. The temperature difference between these two systems at that time is at its minimum, thus accounting for the minimum heat transfer capability.

The heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

4.9.2.1.3 Residual Heat Removal System Valves

Each valve that performs a modulating function is equipped with two stem packing glands and an intermediate leakoff connection that discharges to the drain header.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions.

The suction line from the reactor coolant system is equipped with a pressure relief valve sized to relieve the combined flow of all the charging pumps at the relief valve set pressure. This relief valve is installed to provide overpressure protection for the reactor coolant system under solid plant operations. Each discharge line to the reactor coolant system is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the check valves which separate the residual heat removal system from the reactor coolant system.

The residual heat removal system includes two isolation valves (8701 and 8702) in series in the inlet line between the high pressure reactor coolant system and the lower pressure RHR system. Each isolation valve is interlocked with one of two independent reactor coolant system pressure transmitters. These interlocks prevent the valves from being opened unless the reactor coolant system pressure is less than 425 psig to ensure that the RHR system is not over pressurized. After the valves are open, another set of interlocks will cause the valves to automatically close when the reactor coolant system pressure increases to approximately 585 psig. Many plants have removed this automatic closure in response to recommendation #5 of the 1985 AEOD report referred to in section 4.9.3.

4.9.2.2 System Features and Interrelationships

4.9.2.2.1 Plant Cooldown

The initial phase of reactor cooldown is accomplished by transferring heat from the RCS to the steam and power conversion system via the steam generators. The second phase of cooldown starts with the RHR system being placed in operation. The RHR system is placed in operation approximately four hours after reactor shutdown, when the temperature and pressure of the RCS are approximately 350°F and 425 psig, respectively.

Assuming that two heat exchangers and two RHR pumps are in service, and that each heat exchanger is being supplied with component cooling water at its design flow rate and temperature, the RHR system is designed to reduce the temperature of the reactor coolant from 350°F to 140°F within 16 hours. The heat load handled by the residual heat removal system during the cooldown includes residual and decay heat from the core and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 20 hours following reactor shutdown from extended operations at full power. Coincident with operation of the residual heat removal system, a portion of the reactor coolant flow may be diverted from downstream of the residual heat removal heat exchangers to the chemical and volume control system (CVCS) low pressure letdown line for cleanup and/or pressure control.

Startup of the residual heat removal system includes a warmup period during which the reactor coolant flow through the heat exchangers is limited to minimize thermal shock to the heat exchangers. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the RHR heat exchangers.

The component cooling water is supplied at a constant flow rate to the RHR heat exchangers. The temperature of the return flow can be controlled by manually adjusting the control valves (606, 607) downstream of the heat exchangers coincident with manual adjustment of the heat exchanger bypass valve (HCV-618).

The reactor coolant system cooldown rate is limited by equipment cooldown rates based on allowable stress limits. The available cooldown rate can be affected by the operating temperature limits of the component cooling water system. As the reactor coolant temperature decreases, the reactor coolant flow through the RHR heat exchangers is gradually increased by adjusting the control valve in each heat exchanger outlet line. The normal plant cooldown function of the residual heat removal system is independent of any engineered safety features function.

The normal cooldown return lines are arranged in parallel, redundant flow paths. These lines are also utilized as the low pressure emergency core cooling injection lines to the reactor coolant system. Utilization of the same return lines for emergency core cooling as well as for normal cooldown lends assurance to the proper functioning of these lines for engineered safety features purposes.

4.9.2.2.2 Solid Plant Operations

The residual heat removal system is used in conjunction with the chemical and volume control system (see Figure 4.9-2) during cold shutdown operations (less than 200°F) to maintain reactor coolant chemistry and pressure control. Solid plant operations (no bubble in the pressurizer) is one method of operating the plant during the cold shutdown period. This mode of operation is generally limited to system refill and venting operations. The term "solid plant" refers to the fact that the reactor coolant system is completely filled to the top of the pressurizer with coolant.

The RHR system is used to circulate reactor coolant from the loop 4 hot leg to the cold leg connections on each loop. The RHR system is essentially operating as an extension of the reactor coolant system and is completely filled with reactor coolant. Pressure in the system can be changed by either changing the temperature of the reactor coolant or by varying the mass of the reactor coolant within the system. Varying the temperature of the reactor coolant is not an effective method of RCS pressure control due to the time required to heat the coolant and the large pressure changes that accompany small temperature changes. Volume control of the reactor coolant is preferred because of the faster response and because any desired pressure change can be obtained within controllable limits. Since control of the mass in the RCS is the preferred means of pressure control, a portion of the RHR flow is diverted to the chemical and volume control system through valve HCV-128.

The flow diverted to the CVCS is controlled by the position of the backpressure control valve PCV-131, which is located downstream of the letdown heat exchanger. During solid plant operations the flow water returned to the reactor coolant system is determined by the charging rate, which is controlled through manual positioning of charging flow control valve HCV-182. The chemical and volume control system is also a water-solid system with the exception of the volume control tank, which acts as a buffer or surge volume for the purpose of pressure control. Pressure is controlled by maintaining a constant charging rate and varying the flow rate of the water into the chemical and volume control system (via PCV-131). To maintain a constant pressure in the RCS, both flow rates (charging and letdown), must be equal. If the charging rate exceeds the letdown rate, then the pressure in the RCS will increase. Conversely, pressure in the RCS will decrease if the letdown flow rate exceeds the charging flow rate.

Normally, the backpressure regulating valve, PCV-131, is maintained in the automatic mode of operation and set to control the reactor coolant pressure at a desired setpoint. The volume control tank absorbs any mismatches between the charging and letdown flow rates. Pressure regulation is necessary to maintain the pressure in the RCS to a selected range dictated by the fracture prevention criteria requirements of the reactor vessel.

4.9.2.2.3 Refueling

Both residual heat removal pumps are utilized during refueling to pump borated water from the refueling water storage tank to the refueling cavity. During this operation, the isolation valves in the inlet line from the reactor coolant system (8701 and 8702) are closed, and the isolation valve from the refueling water storage tank (8812) is opened. The reactor vessel head is lifted slightly, and refueling water is pumped into the reactor vessel through the normal RHR system return lines and then into the refueling cavity through the open reactor vessel. The reactor vessel head is gradually raised as the water level in the refueling cavity rises. After the water level reaches the normal refueling level, the reactor coolant system inlet isolation valves are opened, and the refueling water storage tank supply valve is closed.

During refueling, the residual heat removal system is maintained in service, with the number of pumps and heat exchangers in operation as required by the heat load and technical specification minimum flow requirements.

After completion of refueling, the RHR system is used to return the water from the refueling cavity to the refueling water storage tank via manual valve 8735. The water level is drained to the level of the reactor vessel flange. The remainder of the water in the refueling cavity is removed through drains located in the bottom of the refueling canal.

4.9.2.3 System Summary

The residual heat removal system performs both normal plant functions and accident functions. The normal plant function is the transfer of heat from the reactor coolant system to the component cooling water system during shutdown operations. This operation is referred to as the second phase of plant cooldown, which starts when RCS T_{avg} is at 350°F. The RHR system is designed to remove the decay heat associated with the shutdown reactor until the plant is restarted. During the shutdown, if solid plant operations are desired, the RHR system is used in conjunction with the chemical and volume control system for solid plant pressure control.

The RHR system is normally aligned to perform its accident function. During the injection phase following a loss of coolant accident, water is supplied from the refueling water storage tank to the reactor coolant system cold legs. For long-term cooling and recirculation, the RHR system utilizes the containment sump as a source of water, and the RHR heat exchangers to cool the water prior to returning the water to the reactor coolant system.

The RHR system is also used during refueling to remove decay heat and to transfer water between the refueling water storage tank and the refueling cavity.

4.9.2.4 Consequences of Loss of RHR

After the fission process is stopped (i.e., the reactor is shutdown) the continuing radioactive decay of fission products and irradiated core materials produces a significant amount of heat. For a typical 3411-MWt nuclear plant, the power associated with this decay heat is about 20 MWt, 24 hours after shutdown from full power. If a means to remove this heat that is being generated in the core is not available, it is obvious that the temperature of the fuel and fuel cladding will increase. Even if the plant is in a cold shutdown condition, the fuel and clad temperature will continue to increase until the point is reached that clad oxidation and fuel melting can occur.

If the plant is in cold shutdown to perform maintenance or refueling, it is very likely that the RCS will be open with steam generator primary manways removed, the pressurizer relief valves open, the pressurizer safety valves and manways removed, or the reactor vessel head vented. When the plant is in mode 5 (cold shutdown), the technical specifications do not require that containment integrity be maintained. The containment equipment hatch and personnel airlocks could be open, and the positions of containment isolation valves could be indeterminate.

Because of the possibilities for system status and alignment during cold shutdown, the time available to replace lost RCS inventory and to re-establish decay heat removal before bulk boiling, core uncovery and fuel damage takes place will vary from plant to plant. The consequences can be severe because of the inability to contain the radioactive fission products that are released once fuel degradation begins.

4.9.3 NRC and Industry Studies

In addition to the studies being performed in conjunction with the resolution of USI A-45, other studies of decay heat removal capabilities have been conducted by independent NRC and industry nuclear safety groups.

A study published by the Nuclear Safety Analysis Center (NSAC) in 1983, "Residual Heat Removal Experience Review and Safety Analysis" (NSAC Report 52), concludes that the "reliability of shutdown decay heat removal could be an important generic safety issue." The study compiled information on over 250 pressurized water reactor (PWR) events involving RHR systems. Over 100 of the events involved an actual loss or significant degradation of decay heat removal capability when it was required to be operable. The results of the events that had specific safety implications fell into three categories: (1) loss of reactor coolant inventory via the RHR system, (2) overpressurization of the RCS, and (3) loss of long-term decay heat removal capability due to RHR system failures.

Even though loss of RCS inventory during cold shutdown conditions might have previously been thought to be unimportant, the analysis by the NSAC concluded that, in certain instances, the loss of inventory combined with the degraded condition of other systems (permitted by technical specifications) needed to replace the lost RCS coolant demonstrated the potential for core uncovery. In one event, if timely operator action had not been taken, core uncovery could have taken place in about 25 minutes (Sequoyah Unit 1, February 11, 1981).

Because of previous repressurization events that have occurred during cold shutdowns at PWRs, the NRC has required that automatic protective systems to prevent cold overpressure be installed. Improper operation and maintenance of these systems can still render them ineffective. Malfunctions or personnel errors during cold shutdown can result in repressurization of the RCS to the setpoint pressure of the pressurizer code safety valves. High pressures could have significant implications regarding reactor vessel brittle fracture limitations.

Many events have taken place that caused the complete loss of the ability to remove decay heat during shutdown. Even though the majority of the events have taken place long enough after shutdown such that sufficient time existed for recovery, the potential exists for decay heat removal losses that could result in bulk boiling conditions in the core. Coolant boiling could create a significant hazard for personnel working in the area as well as lead to core damage.

The NSAC report concludes that significant improvements in decay heat removal capabilities could be made by simply upgrading plant procedures and administrative controls used during plant shutdown. Historically, utilities have emphasized stringent controls and procedural requirements during power operation. The assumption was that during cold shutdown, the plant was in a “safe” condition and that strict controls and safety equipment operability were not necessary. The results of analysis of repetitive events involving decay heat removal systems have demonstrated that this is not necessarily the case.

Some of the recommendations made in the NSAC report include:

1. Improvements in training and procedures related to loss of RCS coolant during RHR system operation (when automatic ECCS is not required to be available by technical specifications), cold overpressure protection, RCS void formation during cold shutdown, long-term unavailability of the RHR system, restoration of air-bound RHR pumps, and inadvertent automatic RHR system isolation;
2. Better administrative controls for maintenance and surveillance during cold shutdown, vessel level monitoring during partially drained operations, critical valve positioning and status control, outage control by operation personnel, and maintenance prioritization; and

3. Minor hardware modifications including better control room indications and alarms for low RHR system flow, actual valve position, valve controls, and shutdown reactor vessel level monitoring systems, and improved instrumentation, data collection and human engineering for shutdown reactor plant operations.

A case study prepared by the NRC office for Analysis and Evaluation of Operational Data (AEOD), "Decay Heat Removal Problems at U.S. Pressurized Water Reactors" (AEOD/C503), was published in December 1985. This study concludes that "for certain postulated events, unless timely corrective actions are taken, core uncovery could result on the order of one to three hours. To date, no serious damage has resulted from the loss-of-DHR [decay heat removal]-system events that have occurred at U.S. PWRs. However, many of the events which have occurred thus far may serve as important precursors to more serious events."

The study's analysis indicates that the underlying or root causes of most of the loss-of-DHR-system events were related to human-factors deficiencies involving procedural inadequacies and personnel error. The majority of the errors were committed during maintenance, testing, and repair activities in shutdown plants. The leading cause of loss of decay heat removal capability was inadvertent automatic closure of the suction isolation valves as a result of human error.

The results of the AEOD analysis show that, in losses of the DHR system occurring during the early stages of shutdown (e.g., within 24 hours after a reactor trip), with the RCS partially drained, or shortly after activation of the DHR system before the primary system is drained, corrective actions must be taken promptly (i.e., within less than two hours unless a loss of RCS inventory is involved) to either restore the DHR system or to implement alternate methods for removing reactor decay heat. This analysis emphasizes the fact that a loss of decay heat removal capability can lead to a safety-significant event unless timely recovery actions are taken.

The AEOD recommendations for improving the reliability of decay heat removal systems include:

1. Improving human factors by upgrading coordination, planning, and administrative control of surveillance, maintenance, and testing operations which are performed during shutdowns;
2. Providing operator aids to assist in determining the time available for DHR recovery and to assist operators in trending parameters during loss-of-DHR events;
3. Upgrading the training and qualification requirements for operations and maintenance staff;
4. Requiring the use of reliable, well-analyzed methods for measuring reactor vessel level during shutdown modes;
5. Modifying plant design to remove automatic closure interlocks and/or power to the DHR suction isolation valves during periods which do not require valve motion; and

6. Clarifying plant technical specifications to eliminate ambiguities associated with operating mode definitions.

4.9.4 Plant Events

4.9.4.1 Diablo Canyon Unit 2 (4/10/87)

On April 10, 1987, Diablo Canyon Unit 2 experienced a loss of decay heat removal capability in both trains. The reactor coolant system had been drained to the midpoint elevation of the hot-leg piping in preparation for the removal of the steam generator manways. During the 85-minute period that the heat removal capability was lost, the reactor coolant temperature increased from 87°F to the boiling point, steam vented from an opening in the reactor vessel head, water spilled from the partially unsealed manways, and the airborne radioactivity levels in the containment rose above the maximum permissible concentration of noble gases allowed by 10CFR20. The reactor, which was undergoing its first refueling, had been shut down for seven days at the time, and the containment equipment hatch had been opened.

Erroneous level indication, inadequate knowledge of pump suction head/flow requirements, incomplete assessment of the behavior of the air/water mixture in the system, and poor coordination between control room operations and containment activities all contributed to the event. Under the conditions that existed, the system that measured the level of coolant in the reactor vessel indicated erroneously high and responded poorly to changes in the coolant level. In addition, the intended coolant level was later determined to be below the level at which air entrainment due to vortexing was predicted to commence. At the time of the event, the plant staff believed that the coolant level was six inches or more above the level that would allow vortexing.

The event began when a test engineer, in preparation for a planned containment penetration local leak rate test, began draining a section of the reactor coolant pump leakoff return line, which he believed to be isolated. However, because of a leaking boundary valve, this action caused the volume control tank fluid to be drained through the intended test section to the reactor coolant drain tank. The control room operators, who were not aware that the engineer had begun conducting the test procedure, increased makeup flow to stop the level reduction inthe volume control tank. A few minutes later, the operators were informed that the reactor coolant drain tank level was increasing, but they could not determine the source of the leakage. Although the actual level of coolant in the reactor vessel was apparently dropping below the minimum intended level, the indication of level in the vessel remained within the desired control band. Subsequently, the electrical current to the operating RHR pump was observed to be fluctuating. The second pump was started, and the running pump was shut down. The current to the second pump also began to fluctuate, so it was immediately shut down as well.

The operators did not immediately raise the water level in the reactor because they still did not know the source of the leakage, the true vessel level, or the status of the work on the steam generator manways. Operators were sent to vent the RHR pumps. One pump was reported to be vented, and a few minutes later an attempt was made to restart the pump. The electrical current to the motor again began to fluctuate, and the pump was secured. During this period the operators did not know the temperature of the coolant in the reactor vessel because the core-exit thermocouples had been disconnected in preparation for the planned refueling. Within an hour, airborne activity levels in the containment were increasing, and personnel began to evacuate from the containment building.

When the operators learned that the steam generator manways had not been removed, action was initiated to raise the reactor vessel water level by adding water from the refueling storage tank. About 10 minutes later, the test engineer identified the source of the leakage and stopped it. When vessel level had been raised sufficiently, one of the RHR pumps was started, and the indicated pump discharge temperature immediately rose to 220°F. At this time the reactor vessel was slightly above atmospheric pressure, and steam was venting from an opening in the reactor vessel head.

Following the loss of decay heat removal capability at Diablo Canyon, the utility took a number of actions to prevent loss of RHR suction during low level operation and to improve recovery should such a loss occur. These actions included the following: (1) evaluation of the reactor vessel level indicating system to determine the level at which vortexing would occur and the effect of vortexing on level measurement; (2) enhancements of instrumentation to provide accurate level measurement, alarm capability, and core-exit temperature measurement during low level operation; (3) enhancement of procedures to include requirements for verifying proper RHR pump suction before starting the second RHR pump; (4) precautions specifying minimum vessel levels as a function of RHR flow; (5) improvements in work planning, control, and communication to include restriction of the work scope to items that do not have the potential to reduce RCS inventory; and (6) improvement of operator training, including a discussion of the potential causes of RHR flow loss, as well as recovery procedures.

Information Notice 87-23 was subsequently issued by the NRC to alert other licensees to the event, and Generic Letter 87-12 was issued to (1) assess safe operation of PWRs when the reactor coolant system water level is below the top of the reactor vessel; (2) determine whether the RHR system meets the licensing basis of the plant, such as GDC 34 and the technical specifications, in this condition; (3) determine whether there is a resultant unanalyzed event that may have an impact on safety; and (4) determine whether any threat to safety that warrants further NRC attention exists in this condition.

4.9.4.2 North Anna Unit 1 (6/27/87)

On June 21, 1987, North Anna Unit 1 operators discovered that approximately 17,000 gallons of reactor coolant had been lost from the RCS while the unit was in cold shutdown. The delay in discovering the inventory loss resulted from the use of pressurizer level as an indication of reactor coolant inventory, failure to use all available indications, and failure to perform a mass inventory balance.

On June 17, 1987, during preparations for a startup following a refueling outage, a problem developed with a reactor coolant pump motor, requiring removal of the motor.

When the problem was discovered, the unit was at approximately 195°F and 325 psig, with a bubble in the pressurizer. In order to establish plant conditions for removal of the motor (which may involve leakage from the RCS), the plant would normally have been cooled to less than 140°F and drained to the midpoint level of the hot-leg nozzle, and the residual heat removal system would have been placed in operation. In order to expedite the work, the plant was cooled to 110°F, and the pressurizer was cooled by filling the pressurizer while venting it via the power-operated relief valves (PORVs). The pressurizer level was lowered to 80% with the PORVs open. The PORVs were then shut because the vapor-space temperature led the operators to believe that a bubble still existed, and the level was further lowered to 20%. This evolution was conducted in accordance with a procedure that was not specifically intended for draining the system. The operators did not realize that lowering the level with the PORVs shut and then subsequently cooling the pressurizer would cause a vacuum to form in the pressurizer and cause the level to hold at 20%.

On June 18, 1987, the pump motor was uncoupled, and a small amount of expected leakage (estimated at 2 gpm) up the pump shaft was encountered. This leakage was relatively clean water from the seal injection line past the pump seals, which did not provide a tight seal when the motor was uncoupled. Makeup to the RCS was from the volume control tank (VCT). The VCT level was maintained, with the VCT pressure greater than the RCS pressure. The operators believed that maintaining the pressurizer and VCT levels would maintain the reactor coolant inventory by making up for any losses with flow from the VCT to the RCS. Voids consisting of noncondensible gases and vapor formed in the RCS and collected in the system high points (reactor vessel head and steam generator tubes). The voids were not indicated by any decrease in pressurizer level.

On June 21, 1987, a decision was made to reduce the pump shaft leakage by raising the pressurizer level, cycling the PORVs to vent the pressure, and then lowering the pressurizer level to draw a slight vacuum in the pressurizer. This was a condition that already existed, but the operators were unaware of it. When the PORVs were cycled, the pressurizer relief tank pressure dropped, as well as the pressurizer level, indicating that a vacuum already existed in the pressurizer. The reactor vessel level indicating system (RVLIS) indication at this time was 79%; however, the operators were not monitoring this indication because the system had been modified during the previous outage and the operators thought it would be unreliable. Because of the recorder scale and the time span visible on the RVLIS trend recorder, the change in the level indication would only have been noticed by comparing it with a separate plot or by rolling it back 12 to 24 hours to compare it with the present indication. When the condition was discovered, the operators took action to provide makeup to the RCS and to vent the reactor vessel head, as well as to check other available information to account for the system inventory. A total of 17,000 gallons of borated water was required to reestablish the RCS inventory.

The procedure used to establish plant conditions for removing the RCP motor did not contain appropriate instructions for monitoring and maintaining the RCS inventory. The licensee changed the procedure to require a review of the reactor coolant system inventory and routine surveillance of all available level indications, including that from the RVLIS.

4.9.5 Summary

In presenting its proposed resolution of USI A-45 in SECY-88-260, the staff recognized the ongoing actions in implementing the Commission's Severe Accident Policy, one of which was a generic letter to require all plants in operation or under construction to undergo a systematic examination termed the Individual Plant Examination (IPE) to identify any plant-specific vulnerabilities to severe accidents. The IPE analysis is intended to examine and understand the plant emergency procedures, design, operations, maintenance, and surveillance to identify vulnerabilities. The analysis will examine both the DHR systems and those systems used for other functions. It is anticipated that a future extension of the IPE program will require examination of externally-initiated events, some of which significantly contribute to DHR failure-related core damage frequency.

To resolve USI A-45, one of the alternatives proposed by the staff was to have each licensee perform a risk assessment for its plant. This assessment would be done in conjunction with the IPE program. Available options for acceptable risk assessments include performing a Level-1 PRA (enhanced) or performing an analysis using the IDCOR IPEM. Thus, USI A-45 was RESOLVED with the requirement for plant-specific analyses to be conducted under the IPE program.

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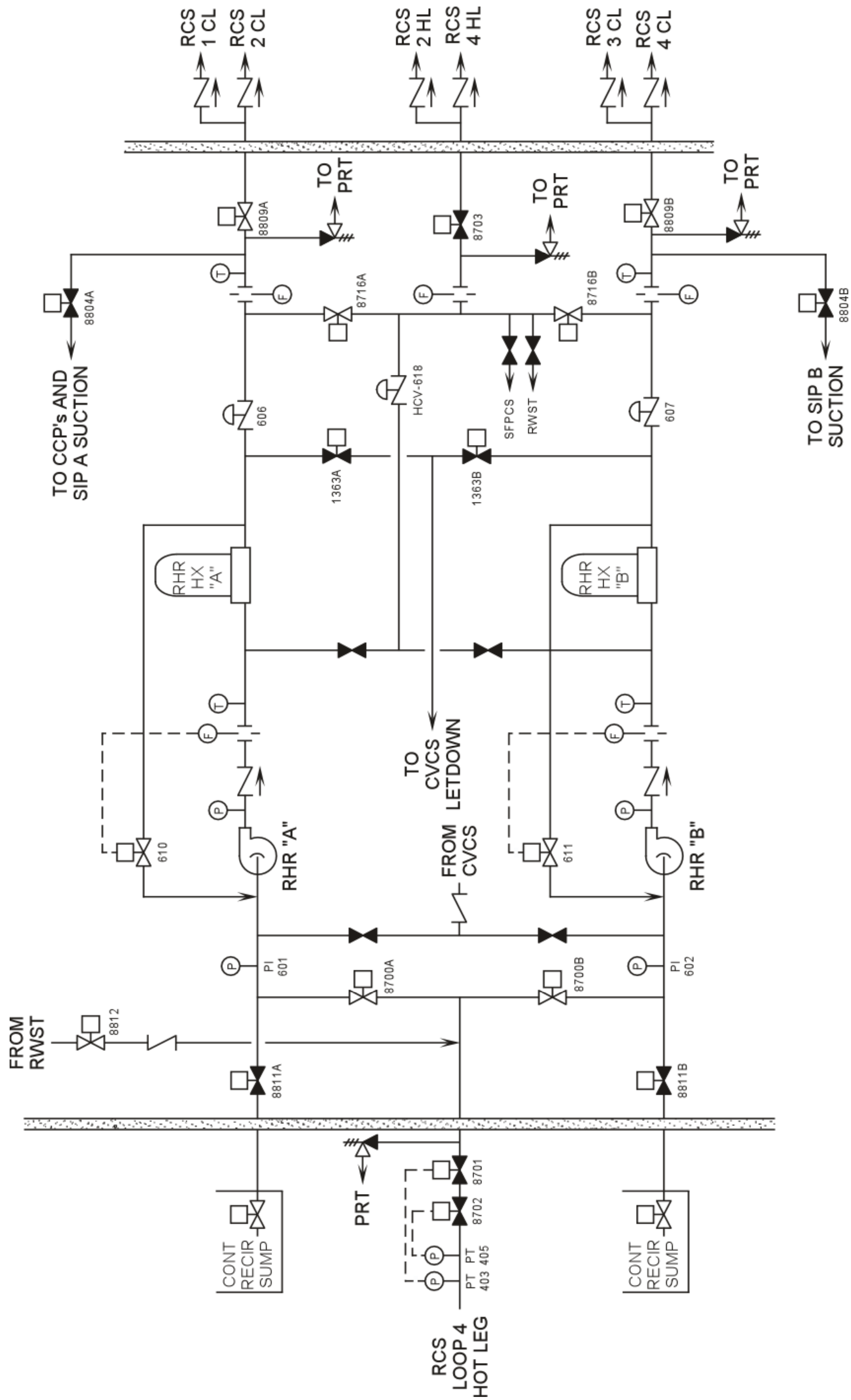

RCS 1 CL
RCS 2 CL
RCS 2 HL
RCS 4 HL
RCS 3 CL
RCS 4 CL
TO PRT
8809A
TO PRT
8703
TO PRT
8809B
8804A
8716A
8716B
8804B
TO CCP's AND SIP A SUCTION
606
HCV-618
SFPCS
RWST
607
TO SIP B SUCTION
1363A
1363B
RHR HX "A"
RHR HX "B"
TO CVCS LETDOWN
610
RHR "A"
FROM CVCS
611
RHR "B"
PI 601
8700A
8700B
PI 602
FROM RWST
8812
8811A
8811B
PRT
8701
8702
PT 403
PT 405
CONT RECIR SUMP
CONT RECIR SUMP
RCS LOOP 4 HOT LEG

Figure 4.9-1 Residual Heat Removal System

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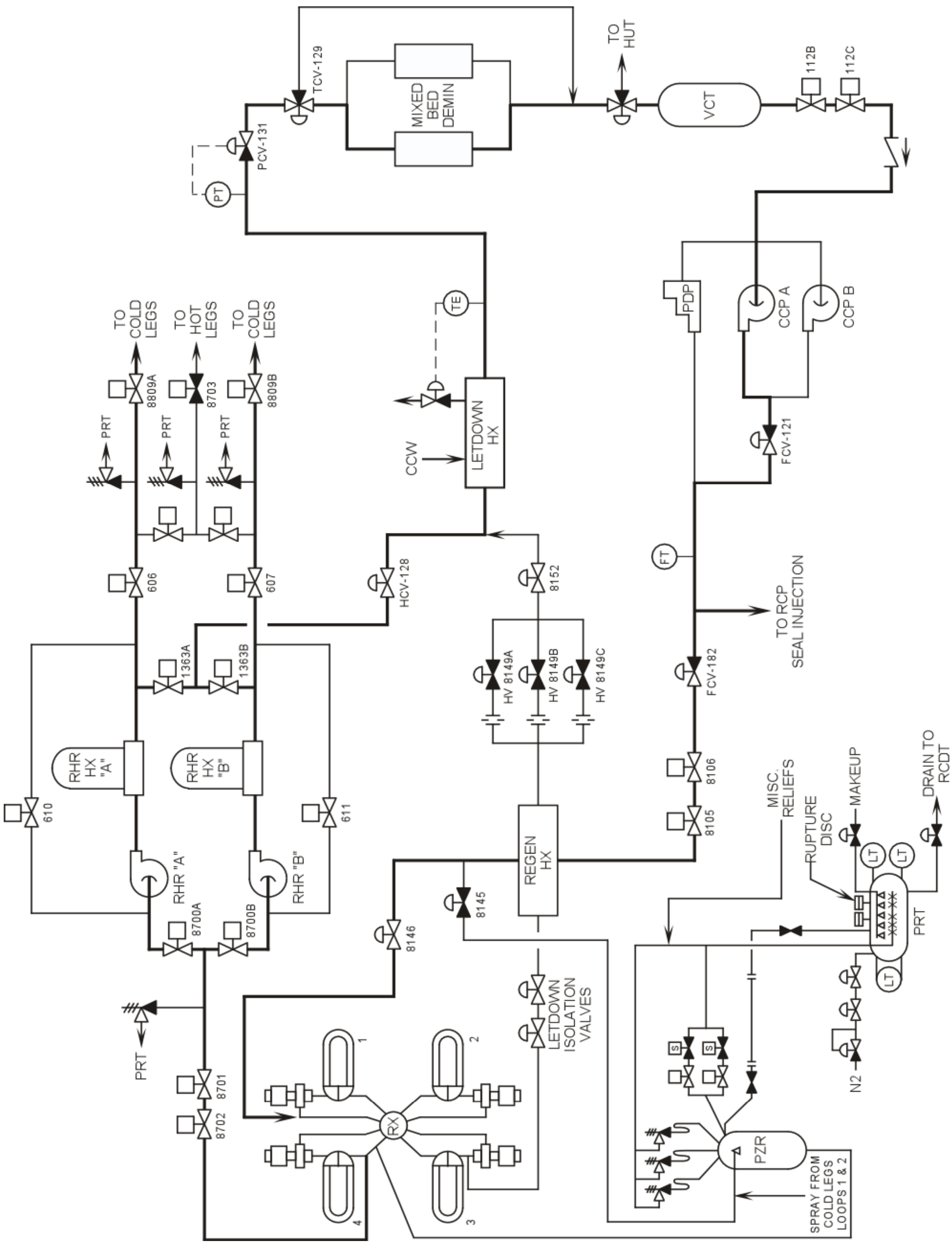

Figure 4.9-2 Solid Plant Pressure Control

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